January 18, 2006

Mr. Dennis L. Koehl
Site Vice President
Point Beach Nuclear Plant
Nuclear Management Company, LLC
6590 Nuclear Road
Two Rivers, WI 54241-9516

SUBJECT: POINT BEACH NUCLEAR POWER PLANT, UNITS 1 AND 2 , NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000266/2005018; 05000301/2005018 (DRS)

Dear Mr. Koehl:

On December 16, 2005, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection of the Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the Point Beach Nuclear Power Station. The enclosed report documents the results of the inspection, which were discussed with Mr. D. Koehl and others of your staff at the completion of the inspection on December 16, 2005.

The inspectors examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel. Based on the results of the inspection, two NRC-identified findings and one self-revealing finding of very low safety significance were identified which involved violations of NRC requirements. However, because these violations were of very low safety significance and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a Non-Cited Violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Prairie Island Nuclear Generating Plant facility.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

Docket Nos. 50-266; 50-301
License Nos. DPR-24; DPR-27

Enclosure: Inspection Report 05000266/2005018; 05000301/2005018 (DRS)

cc w/encl: F. Kuester, President and Chief
Executive Officer, We Generation
J. Cowan, Executive Vice President
Chief Nuclear Officer
D. Cooper, Senior Vice President, Group Operations
J. McCarthy, Site Director of Operations
D. Weaver, Nuclear Asset Manager
Plant Manager
Regulatory Affairs Manager
Training Manager
Site Assessment Manager
Site Engineering Director
Emergency Planning Manager
J. Rogoff, Vice President, Counsel & Secretary
K. Duveneck, Town Chairman
Town of Two Creeks
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Public Service Commission of Wisconsin
J. Kitsembel, Electric Division
Public Service Commission of Wisconsin
State Liaison Officer

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-266; 50-301
License No:	DPR-24; DPR-27
Report No:	05000266/2005018; 05000301/2005018 (DRS)
Licensee:	Nuclear Management Company, LLC
Facility:	Point Beach Nuclear Power Plant
Location:	6590 Nuclear Road Two Rivers, WI 54241-9516
Dates:	December 12 through 16, 2005
Inspectors:	R. Daley, Senior Reactor Inspector, Team Leader C. Acosta, Reactor Inspector B. Jose, Reactor Inspector C. Moore, Operations Engineer (Observer) N. Valos, Senior Operations Engineer
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety (DRS)

SUMMARY OF FINDINGS

IR 05000266/2005018; 05000301/2005018 (DRS); 12/12/2005 - 12/16/2005; Point Beach Nuclear Power Station, Units 1 and 2; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59) and Permanent Plant Modifications.

The inspection covered a one-week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by four regional based engineering inspectors. Three Green Non-Cited Violations (NCV) were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red), using Inspection Manual Chapter 0609, "Significance Determination Process (SDP.)" Findings for which the SDP does not apply, may be Green, or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

Green. The inspectors identified a Severity Level IV Non-Cited Violation of 10 CFR 50.59(d)(1) for the licensee's failure to perform a safety evaluation for compensatory actions taken for an activity associated with a degraded plant condition. Specifically, the licensee "screened out" an activity which replaced an automatic action for Chemical and Volume Control System (CVCS) letdown isolation on low pressurizer level with a manual action to isolate letdown on low pressurizer level, while replacing the Unit 2 pressurizer low level bistables with Unit 2 online at power. At the end of the inspection period, the licensee planned to perform a safety evaluation in accordance with 10 CFR 50.59 for the compensatory actions taken for the activity associated with the degraded plant condition.

Because the issue affected the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors, at the time of the inspection, could not reasonably determine that the UFSAR change, which adversely affected equipment important to safety, would not have ultimately required NRC approval. The inspectors evaluated the finding using IMC 0609, Appendix A, Phase 1 screening for the mitigating systems cornerstone and determined that the finding was of very low safety significance because the finding was not a design or qualification deficiency that was confirmed to result in a loss of operability or functionality per "Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment." (Section 1R02.1.b.2)

Green. A self-revealed finding of very low safety significance was associated with a violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control." During replacement of the Service Water outlet valves for the Component Cooling Water (CCW) heat exchangers, the licensee failed to evaluate design differences between the original valves and the replacement valves. These differences led to the eventual failure of the stems in both valves. This issue was entered into the licensee's corrective action system.

The issue was more than minor because it affected the mitigating system cornerstone attribute of “Design Control” and affected the cornerstone objective of ensuring reliability of systems that respond to initialing events to prevent undesirable consequences. Specifically, failure of these valves could prevent proper cooling of safety related systems. The finding screened as having very low significance (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situations,” because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. While the design deficiency led to failure of the valves, the failures occurred during a plant shutdown; therefore, the valves would not have been required to function as designed. (Section 1R17.1.b.1)

Cornerstone: Barrier Integrity

Green. The inspectors identified a Severity Level IV Non-Cited Violation associated with the failure to perform an adequate safety evaluation review as required by 10 CFR 50.59 for changes made to the facility as described in the UFSAR. In safety evaluation, EVAL 2004-003, the licensee failed to provide a basis for the determination that on-line repairs to the excess letdown line with a freeze seal in place as a boundary for Reactor Coolant System (RCS) effluent from the Reactor Coolant Pumps (RCPs) was acceptable without a license amendment. Specifically, for this freeze seal evolution, the licensee would have replaced the American Society of Mechanical Engineers (ASME) Class II, Seismic Class I piping in the excess letdown line with a freeze plug while the plant was still on-line. Within the 10 CFR 50.59 evaluation, the licensee failed to provide a basis for why this freeze seal evolution did not present more than a minimal increase in the likelihood of occurrence of a malfunction of a Structure, System and Component (SSC) important to safety. As a result of this issue, the licensee performed a revision to the original safety evaluation to withdraw the facility change that allowed the freeze seal with the plant online.

Because the issue affected the NRC’s ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because the inspectors could not reasonably determine that the UFSAR change, which adversely affected equipment important to safety, would not have ultimately required NRC approval. The finding was determined to be of very low safety significance (Green), because the inspectors answered “no” to all three questions under the Containment Barriers Cornerstone column of the Phase 1 worksheet. Specifically, the licencee had not actually performed this evolution when the pressure boundary was required to be intact. (Section 1R02.1.b.1).

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

From December 12 through 16, 2005, the inspectors reviewed six evaluations performed pursuant to 10 CFR 50.59. The inspectors confirmed that the evaluations were thorough and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 14 screenings where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. The list of documents reviewed by the inspectors is included as an attachment to this report.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

b. Findings

b.1 Updated Final Safety Analysis Report Change to Replace ASME Class II, Seismic Class I, Piping with a Freeze Seal

Introduction: The inspectors identified a Severity Level IV Non-cited Violation (NCV) of very low safety significance for failing to perform an adequate safety evaluation in accordance with 10 CFR 50.59. The safety evaluation, EVAL 2004-003, involved an updated Final Safety Analysis Report (UFSAR) change that allowed on-line repairs to the excess letdown line with a freeze seal in place as a boundary for RCS effluent from the RCPs. Within the safety evaluation, the licensee failed to provide a basis for why this freeze seal evolution did not present more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety.

Description: The licensee initiated 10 CFR 50.59 Evaluation 2004-003 to help facilitate repairs to valve 2CV-285, "Excess Letdown Outlet Valve." To perform this work with the plant on-line, the licensee proposed to install a freeze seal upstream of 2CV-285 to establish isolation of the valve from the RCP seal return line. The line that the freeze seal would be placed on was qualified as ASME Class II, Seismic Category I, pressure boundary piping. During the work, the bonnet for 2CV-285 would be removed; however, if the freeze seal were to fail, a contingency action was put in place for operators to manually realign 3-way valve 2CV-312, located upstream of the freeze seal, so that water could be diverted to the Reactor Coolant Drain Tank (RCDT), and thus minimize leakage.

The inspectors noted that the 10 CFR 50.59 evaluation did not address the freeze seal evolution properly in its response to the question that asked if the activity resulted in a more than minimal increase in the likelihood of occurrence of a malfunction of a Structure System or Component (SSC) important to safety previously evaluated in the current license basis. The 10 CFR 50.59 evaluation contained little justification for the downgrading of the ASME Class II pressure boundary and no justification for the downgrading of the Seismic Class I qualification. The freeze plug did not provide the same degree of barrier integrity as an ASME Class II, Seismic Class I, pipe. The inspectors were concerned, because the licensee would be substituting an ASME Class II, Seismic Category I, boundary with a non-Code recognized pressure boundary material (e.g., ice plug). This was not a permissible repair under Section XI, Article IWA-4000, of the ASME Code, which only recognizes pressure boundary repair by welding, brazing, or metal removal. The safety function of the line was to serve as a pressure boundary for reactor coolant from the RCP seal return line, while the malfunction would be a line break. In response to this question in the 10 CFR 50.59 evaluation, the licensee justified the evolution primarily by stating that strong procedural controls were in place to prevent any anticipated problems. This approach was insufficient to show that downgrading of the boundaries would not have result in more than a minimal increase in the likelihood of occurrence of a malfunction of the pressure boundary line.

Although the facility change described in the UFSAR and allowed by the 10 CFR 50.59 evaluation allowed this evolution to occur on-line, the licensee had instead performed the evolution during an outage. While this was fortuitous, the licensee had still changed the facility as described in the UFSAR to allow this type of evolution.

The inspectors determined that the 10 CFR 50.59 evaluation that was performed to allow this freeze seal evolution with the plant on-line was not in accordance with the requirements in 10 CFR 50.59, because the evaluation did not adequately address the downgrading of the pressure boundary from ASME Class II to a freeze plug, and because it did not address the deletion of the Seismic Class I qualification for the line. The inspectors noted that this change to the UFSAR may have resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety, since the freeze plug would not have provided the same degree of barrier integrity for the RCS effluent from the RCP seals as the actual ASME Class II, Seismic Class I, piping would have. The licensee entered this condition into their corrective action program as CAP069372. As a result, the licensee performed a revision to the original 10 CFR 50.59 evaluation to withdraw the original facility change that allowed this freeze seal evolution with the plant on-line.

Analysis: The inspectors determined that this issue was a performance deficiency, since the licensee permanently changed the facility as described in the UFSAR without providing the necessary justification under 10 CFR 50.59 for the reduction of the pressure boundary of the excess letdown line from an ASME Class II, Seismic Class I, pipe to a freeze seal. The finding was determined to be more than minor, because the inspectors could not reasonably determine that the UFSAR change, which adversely affected equipment important to safety, would not ultimately have required NRC approval.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the Significance Determination Process (SDP). However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the finding screened as having very low significance (Green) using IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," because the inspectors answered "no" to all three questions under the Containment Barriers Cornerstone column of the Phase 1 worksheet. Specifically, the licensee had not actually performed this evolution when the pressure boundary was required to be intact. Based upon this Phase 1 screening, the inspectors concluded that the issue was of very low safety significance (Green). In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation.

Enforcement: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written evaluation which provides a basis for the determination that the change, test, or experiment does not require a license amendment.

Contrary to the above, in safety evaluation, EVAL 2004-003, the licensee failed to provide an adequate basis for the determination that on-line repairs to the excess letdown line with a freeze seal in place as a boundary for RCS effluent from the RCPs was acceptable without a license amendment. Specifically, for this freeze seal evolution, the change in the UFSAR, dated March 29, 2004, allowed replacement of the ASME Class II, Seismic Class I, piping in the excess letdown line with a freeze plug while the plant was still on-line. Within the 10 CFR 50.59 evaluation, the licensee failed to provide a basis for why this freeze seal evolution did not present more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety. In accordance with the Enforcement Policy, this violation of the requirements of 10 CFR 50.59 was classified as a Severity Level IV Violation because the underlying technical issue was of very low safety significance. Because this non-willful violation was non-repetitive, and was captured in the licensee's corrective action program (CAP069372), it is considered a Non-Cited Violation consistent with VI.A.1 of the NRC Enforcement Policy (NCV). (NCV 05000266/ 2005018-01; 05000301/2005018-01 (DRS))

b.2 Failure to Perform a 10 CFR 50.59 Evaluation for Compensatory Actions Associated with Letdown Line Automatic Isolation

Introduction: The inspectors identified a Severity Level IV Non-Cited Violation of very low safety significance for failing to perform a safety evaluation in accordance with 10 CFR 50.59(d)(1) for the compensatory actions taken for an activity associated with a degraded plant condition. Specifically, the licensee "screened out" an activity which replaced an automatic action for Chemical and Volume Control System (CVCS) letdown isolation on low pressurizer level of 12 percent with a manual action to isolate letdown if pressurizer level decreased to 20 percent, while replacing the Unit 2 pressurizer low level bistables with Unit 2 online at power.

Description: On February 5, 2004, a 10 CFR 50.59 screening (SCR 2004-0031) was completed to evaluate an activity to replace the Unit 2 pressurizer level low bistables online while preventing CVCS letdown isolation from occurring and maintaining the pressurizer backup heaters in service. The licensee planned to replace the pressurizer low level bistables due to an inadvertent loss of Unit 2 letdown that occurred on February 4, 2005. The licensee attributed the probable cause of the inadvertent loss of letdown to a spurious failure of a pressurizer low level bistable. The licensee's 10 CFR 50.59 screening for the activity concluded that a 10 CFR 50.59 evaluation was not required because, in part, the activity did not adversely affect a design function. The 10 CFR 50.59 screening stated that a designated operator would be stationed to monitor pressurizer level and would manually initiate letdown isolation and deenergize the pressurizer backup heaters if pressurizer level dropped below 20 percent. The activity to replace the Unit 2 pressurizer low level bistables was completed on February 6, 2005.

The functions of the pressurizer low level bistables that were replaced were to isolate CVCS letdown and deenergize the pressurizer backup heaters on a low pressurizer level of 12 percent. The design function to isolate CVCS letdown on low pressurizer level was addressed in the Point Beach Nuclear Plant (PBNP) UFSAR Table 9.3-7, "Malfunction Analysis of Chemical and Volume Control System." As stated in FSAR Table 9.3-7, the letdown isolation function prevented supplementary loss of coolant during a letdown line rupture event inside the reactor containment.

The inspectors noted that guidance contained in Section 4.4 of Nuclear Energy Institute Standard NEI 96-07, "Guidelines for 10 CFR 50.59 Evaluations," Revision 1, which the NRC endorsed in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," stated, in part, that if interim compensatory actions are taken to address a degraded condition, 10 CFR 50.59 should be applied to determine whether the compensatory actions impact aspects of the facility described in the UFSAR. In this case, the substitution of manual actions for the letdown isolation function was an adverse change that warranted a full 10 CFR 50.59 evaluation.

The inspectors concluded and the licensee subsequently concurred that the activity to replace the pressurizer low level bistables required a safety evaluation. At the end of the inspection period, the licensee planned to perform a safety evaluation in accordance with 10 CFR Part 50.59 for the compensatory actions taken for the activity associated with the degraded plant condition.

Analysis: The inspectors determined that the licensee's failure to perform a 10 CFR 50.59 evaluation for this substitution of manual actions for automatic actions was a licensee performance deficiency warranting a significance evaluation. This finding was determined to be more than minor because the inspectors could not reasonably determine that the change would not ultimately have required NRC approval.

Because violations of 10 CFR 50.59 are considered to be violations that potentially impede or impact the regulatory process, these violations are dispositioned using the traditional enforcement process instead of the SDP. However, if possible, the underlying technical issue is evaluated under the SDP to determine the severity of the violation. In this case, the inspectors determined that even though the change was not adequately evaluated in accordance with 10 CFR 50.59, this violation was of very low safety significance, because the design function of mitigating systems to respond to this initiating event scenario were not adversely affected. The inspectors evaluated the finding using IMC 0609, Appendix A, Phase 1 screening for the mitigating systems cornerstone and determined that the finding was of very low safety significance because the finding was not a design or qualification deficiency that was confirmed to result in a loss of operability or functionality per "Part 9900, Technical Guidance, Operability Determination Process for Operability and Functional Assessment."

Enforcement: Title 10 CFR 50.59(d)(1) states, in part, that the licensee shall maintain records of changes in the facility, of changes in procedures, and of tests and experiments. These records must include a written evaluation which provides the bases for the determination that the change, test, or experiment does not require a license amendment.

Contrary to the above, the licensee failed to perform a written safety evaluation to address compensatory actions associated with an activity which replaced an automatic action for CVCS letdown isolation on low pressurizer level with a manual action to isolate letdown on low pressurizer level, while replacing the Unit 2 pressurizer low level bistables with Unit 2 online at power. The results of this violation were determined to be of very low safety significance; therefore, this violation was classified as a Severity Level IV Violation of 10 CFR 50.59. Because this violation was of very low significance, non-willful, non-repetitive, and documented in the licensee's corrective action program as CAP069337, this finding is being treated as a Non-Cited Violation (NCV), consistent with Section VI.A. of the NRC Enforcement Policy. (NCV 05000266/2005018-02; 05000301/2005018-02 (DRS))

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From December 12 through 16, 2005, the inspectors reviewed nine permanent plant modifications that had been installed in the plant during the last two years. The modifications were chosen based upon risk significance, safety significance, and complexity. As per inspection procedure 71111.17B, one modification was chosen that affected the barrier integrity cornerstone. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements and the licensing bases and to confirm that the changes did not adversely affect any systems' safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an attachment to this report.

b. Findings

b.1 <u>Failure to Apply Adequate Design Controls During Replacement of Service Water (SW) Valves SW-360 and SW-322</u>

Introduction: During replacement of the SW outlet valves for the CCW heat exchangers, the licensee did not implement adequate design controls in accordance with 10 CFR 50, Appendix B, Criterion III. Specifically, the licensee failed to evaluate design differences between the original valves and the replacement valves. These differences led to the eventual failure of the stems in both valves. This self-revealing finding was considered to be of very low safety significance and was dispositioned as a Green NCV.

Description: Spare Parts Equivalency Evaluation Documents (SPEED) 2005-079 and 2005-080 were written to replace SW valves SW-360 and SW-322, respectively. These two valves served as the CCW heat exchanger outlet valves for the service water cooling.

The SPEEDs were written to evaluate the change, since a new type valve was being put in to replace the older valves. Under the Point Beach SPEED process, this change was considered an alternate replacement. As a part of the process for this type of replacement, the licensee was required to justify the differences between the old valves and the new valves. In this justification, the licensee compared the plug and seat design of the valves noting that the old valve contained a web and bridge design as well as a valve plug with a spindle at the end of it. The spindle was designed to slide through the bridge, ensuring proper alignment and uniform seating in all directions. The new design did not have these features. This was a very important parameter, because the flow of water through the valve could be excessive in this system. The design of the old valve helped reduce the amount of vibration that the valve parts experienced during valve operation. While the SPEED evaluations mentioned this design difference, it did not provide a justification for the absence of these design features in the new valve.

As documented in the licensee’s corrective action document (CE016479), because of these design differences, the new valves were not well suited for throttling at the flow rate seen in the application. Consequently, the valve stems for both valves broke approximately a week after installation, after being placed back inservice, but before returning the plant to operation.

Analysis: The inspectors determined that this self-revealed failure to assure design controls commensurate with the valves’ original design was a performance deficiency warranting a significance determination. Specifically, the licensee changed valve designs for the Service Water outlet valves from the CCW heat exchangers and did not evaluate the effects of those design changes. This failure to fully evaluate these effects resulted in the installation of an inadequate design that could not withstand the flow of the system resulting in the breaking of the valve stem for both valves.

The issue was more than minor because it was associated with the Mitigating System cornerstone attribute of “Design Control,” and affected the cornerstone objective of ensuring reliability of systems that respond to initiating events to prevent endesirable consequences. Specifically, failure of these valves could prevent proper cooling of safety related systems. The finding screened as having very low significance (Green) using IMC 0609, Appendix A, “Significance Determination of Reactor Inspection Findings for the At-Power Situations,” because the inspectors answered “no” to all five questions under the Mitigating Systems Cornerstone column of the Phase 1 worksheet. While the design deficiency led to failures of the valves, the failures occurred during a plant shut-down, therefore, the valves would not have been required to function as designed.

Enforcement: 10 CFR Part 50, Appendix B, Criterion III, "Design Control" states, in part, that design changes, including field changes, shall be subject to design control measures commensurate with those applied to the original design. Contrary to the above, during the replacement of Service Water valves SW-360 and SW-322, the licensee did not evaluate the design differences between the original valves and the new valves for this field change as their design process required. This resulted in the eventual stem breakage of these valves.

Because this failure to apply appropriate design control measures was determined to be of very low safety significance and because it was entered in the licensee's corrective action program as CAP068445, this violation is being treated as an NCV, consistent with Section VI.A of the NRC Enforcement Policy. (NCV 05000266/2005018-03; 05000301/2005018-03 (DRS))

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From December 12 through 16, 2005, the inspectors reviewed twelve Corrective Action Process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4. OTHER ACTIVITIES

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Mr. D. Koehl and others of the licensee's staff, on December 16, 2005. Licensee personnel acknowledged the inspection results presented. Licensee personnel were asked to identify any documents, materials, or information provided during the inspection that were considered proprietary. No proprietary information was identified.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee

C. Butcher, Engineering Director
K. Dittman, Supervisor - Electrical Engineering
R. Grazio, Regulatory Affairs Manager
D. Koehl, Site Vice President
J. McNamara, Supervisor - Mechanical Design
L. Peterson, Engineering Design Manager
L. Schofield, Senior Engineer - Regulatory Affairs
S. Scott, Senior Engineer - Design Engineering

Nuclear Regulatory Commission

B. Burgess, Reactor Projects Branch 2
H. Chernoff, NRR
D. Hills, Chief, Engineering Branch 1
R. Krsek, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

None.

Opened and Closed

05000266/2005018-01; 05000301/2005018-01	NCV	Updated Final Safety Analysis Report Change to Replace ASME Class II, Seismic Class I, Piping with a Freeze Seal
05000266/2005018-02; 05000301/2005018-02	NCV	Failure to Perform a 10 CFR 50.59 Evaluation for Compensatory Actions Associated with Letdown Line Automatic Isolation
05000266/2005018-03; 05000301/2005018-03	NCV	Failure to Apply Adequate Design Controls During Replacement of Service Water (SW) Valves SW-360 and SW-322

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather, that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

IR02 Evaluation of Changes, Tests, or Experiments (71111.02)

10 CFR 50.59 Screenings

SCR-2002-0384-01; Auxiliary Feedwater Pump Room Fire Wall Addition; dated February 13, 2004

SCR 2004-0031; Defeating PZR Low Level Letdown and Heater Cutoff to Replace Bistables Online; dated February 5, 2004

SCR 2004-0091; OP 4F; Revision 0; Reactor Coolant system Reduced Inventory Requirements and Nozzle Dam Operational Requirements - Unit 1 and Unit 2; dated April 21, 2004

SCR 2004-0093; Temporary Change to Become Permanent to HPIP 7.51.6; Isolation of the Containment Ventilation System Using the RMS High Alarm Automatic Trip Functions; Revision 14; TCN 2004-0339; Revision 00

SCR 2004-0093-01; Temporary Change to Become Permanent to HPIP 7.51.6, Isolation of the Containment Ventilation System Using the RMS High Alarm Automatic Trip Functions; Revision 14; TCN 2004-0339; Revision 01

SCR 2004-0138; Unit 1 RHR Cross Connect Valves Procedural Closing Requirements; dated May 31, 2004

SCR-2004-0195; Revision to IT-72; Service Water Valves; Attachment B (oSW-02817); dated January 15, 2004

SCR 2004-0261; Units 1 and 2 EOP-3; Steam Generator Tube Rupture; Revision 36; dated October 14, 2004

SCR 2005-0081; Change Battery Restoration Times in DC SOPs; dated July 27, 2005

SCR 2005-0101; EDG Fuel Oil Duplex Filter Operations; dated April 16, 2005

SCR 2005-0175-01; Revise 1P-15B, 2P-15A, and 2P-15B Safety Injection Pump Motor Over Current Relay Setpoints; dated August 2, 2005

SCR 2005-0189; New Procedure 1-sop-y06 to Provide Operational Guidance for Removing from and Restoring to Service Instrument Panel 1-Y06; dated August 9, 2005

SCR 2005-0192; Revision to FSAR 6.4 to Correct an Error and Remove Specific Procedural Details; dated September 6, 2005

SCR 2005-0194; Revision 26 to Procedure AOP-0.0; "Vital Dc System Malfunction"; dated August 31, 2005

SCR 2005-0200; New Procedure 1-SOP-4KV-001 for Islanding 1A05 & 1A06 Buses to Their Respective Emergency Diesels; dated August 30, 2005

10 CFR 50.59 Evaluations

SE 2004-001; Revise Time to Reach RHR Cut In Conditions in Various FSAR Radiological Consequences Evaluations; dated August 22, 2005

EVAL 2004-003; Evaluation to Support TM 04-006 for Repairs to 2CV-285, "Excess Letdown Outlet Valve"; dated March 29, 2004

EVAL 2004-004; MR 03-041; Repair of Unit 1 Reactor Vessel Head CRD Penetrations; dated May 13, 2004

EVAL 2004-004-01; MR 03-041 – Repair of Unit 1 Reactor Vessel Head CRDM Penetrations; dated May 26, 2004

SE 2005-003; MR 99-035*A/B, MR 99-036*A/B - Containment Hatch Airlock Equalizing Valve Replacement; dated October 13, 2005

EVAL 2005-007; Revision to PC 29; "Monthly Gas Turbine and Auxiliary Diesel Load Test," Oi 110; "Gas Turbine Operation and Np-2.1.5, "Electrical Communications Switchyard Access and Work Planning"; dated August 26, 2005

IR17 Permanent Plant Modifications (71111.17B)

Modifications

MR 99-036*A; Upgrade Unit 2 Containment Airlock Operating Mechanism (C-2); dated April 6, 2004

MR 99-036*B; Upgrade Unit 2 Containment Airlock Operating Mechanism (C-1); dated April 6, 2004

MR 00-037; Replacement of DC Breakers in Main Control Boards; dated March 22, 2004

MR 01-063; Replace Service Water Pump Motor On P-32B to Improve Reliability; dated April 10, 2003

MR 01-074; Add Time Delays to Battery Charger and Dc Bus Voltage Alarm Circuits; dated June 21, 2002

MR 02-011*B; Extend Unit 1 SI and RH High Point Vent Lines; dated March 27, 2004

MR 01-144; AFW Motor Driven Pump Mini Recirc Control Valve Modification; dated December 11, 2001

SPEED 2005-079; Replacement of 12 inch Powell Globe Valve (SW-360); dated September 27, 2005

SPEED 2005-080; Replacement of 12 inch Powell Globe Valve (SW-322); dated September 27, 2005

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

CAP069337; SCR 2004-0031 Should Have Resulted in a 50.59 Evaluation; dated December 13, 2005

CAP069343; Appropriateness of Change to EOP 3.0 in Question; dated December 13, 2005

CAP069365; Background and Deviation Documents are Not Current with EOP-3 Revision 37; dated December 14, 2005

CAP069385; Calculation Documentation Deficiency; dated December 15, 2005

CAP069391; ISI Classification Error for SI High Point Vent Tubing; dated December 15, 2005

CAP 069398; Calculation Project HVAC – Explicitly Address Uncertainty and Heat Sink Info; dated December 15, 2005

PCR027728; Revise RMP 9225-2 to Reference Tech Spec Surveillance SR 3.6.2.2; dated December 13, 2005

Corrective Action Program Documents Reviewed During the Inspection

CAP 001618; All A-Train Diesel Fuel Oil Pump Power Lost During Postulated Fire; dated April 23, 1999

CAP053555; Unit 2 Inadvertent Letdown Isolation; dated February 4, 2004

CAP055833; Conflict in TS SRs for LCO 3.9.3; dated April 17, 2004

CAP056416; 1RH-713B RHR Pump Discharge Cross Connect Does Not Isolate Per Design; dated May 5, 2004

CAP 057089; Modification to Unit 1 Feedwater Piping Not Done as Required; dated May 28, 2004

CAP061764; Interim Condition Existing for Greater Than 90 Days Without 50.59 Screening/Evaluation; dated January 28, 2005

CAP063023; Inadequate 10CFR50.59 Screening for AOP 0.0 Temporary Change 2005-0012; dated March 24, 2005

CAP 066419; Conclusion of 50.59 screening SCR 2002-0377 questioned by NRC; dated August 16, 2005

CAP068445; A/B CCW HX SW Outlet Vibration; dated October 30, 2005

CAP068529; 1SW-322 Difficult to Operate; dated November 2, 2005

CAP068622; Replacement for 1SW-322 and 1SW-360 Mechanically Failed after Installation; dated November 4, 2005

CAP068674; 1HX-12A Component Cooling Water Heat Exchanger Needs to Be Returned to Service; dated November 7, 2005

Calculations

Calculation 692301-2.2-004-00-A; AFW Pump Room Loss of HVAC Analysis; dated January 29, 1990

Addendum to Calculation 692301-2.2-004-00-A; AFW Pump Room Loss of HVAC Analysis; dated August 28, 2003

Calculation 2002-0002; Nitrogen Backup System for MDAFP Discharge Valves (AF-4012/4019) and Minimum Flow Recirculation Valves (AF 4007/4014); Revision 3

Drawings

Drawing 290585; Fire Protection for Turbine Building, Aux Building and Containment Elevation 8’ 00’’; Revision 16

Procedures

AOP-5B; Loss of Instrument Air; Revision 27

BG-EOP-3; Steam Generator Tube Rupture; Revision 31

CL 7A; Safety Injection Checklist Unit 1; Revision 23

CL 7B; Safety Injection Checklist Unit 1; Revision 21

EOP-3 Unit 1; Steam Generator Tube Rupture; Revision 37

IT 03E; Manual Stroke of Low Head Safety Injection Valves (Quarterly) Unit 1; Revision 7

NMC 50.59 Resource Manual; Section 5.0; The 10 CFR 50.59 Screening; Revision 2

NMC 50.59 Resource Manual; Section 6.0; 10 CFR 50.59 Evaluation; Revision 2

NP 2.1.4; Operator Burdens; Revision 5

NP 5.1.8; 10 CFR 50.59/72.48 Applicability, Screening, and Evaluation; Revision 6

NP 7.2.15; Fleet Modification Process; Revision 6

NP 7.2.25; Modification Turnover and Closeout; Revision 0

OI 128; SI System Fill and Vent Unit 1; Revision 11

OI 135A; Fill and Vent Train A RHR System Unit 1; Revision 8

OI 135B; Fill and Vent Train B RHR System Unit 1; Revision 10

OM 3.26; Use of Dedicated Operators; Revision 9

OP 4G; Steam Generator Nozzle Dam Operational Requirements Unit 1; Revision 0

OP 7B; Removing Residual Heat Removal System from Operation; Revision 37

PBF-2032; Daily Log Sheet; Revision 80

RMP 9225-2; Defeating/Restoring Containment Personnel and Escape Hatch Door Interlocks; Revision 7

1TS-ECCS-002; Safeguards System Venting (Monthly) Unit 1; Revision 6

<u>Miscellaneous Documents</u>

Calculation 2001-0024; Containment Airlock and Door Seal Pressure Testing Acceptance Criteria; Revision 3

EOPSTPT T.1; RCP Trip; Revision 0

Engineering Evaluation 2004-0006; Effect of AFW Appendix R Firewall on Room Heatup Due to Loss of HVAC Calculations; dated February 19, 2004

Modification 02-029; Aux Feed Mini Recirc Safety Upgrade/Remove AF-117 Internals; dated August 20, 2002

Completed OI 92A; Fuel Oil Ordering, Receipt Sampling and Offloading; dated April 4, 2005

Completed OI 92A; Fuel Oil Ordering, Receipt Sampling and Offloading; dated December 7, 2005

Operations Work Plan 2004-033; 1RH-713A and B Torque Determination; dated May 31, 2004

NRC SER dated July 9, 1997; Safety Evaluation Related to Amendment Nos. 174 and 178 to Facility Operating License Nos DPR-24 and DPR-27; dated July 9, 1997

Completed PBF 3005; Blended #1 and #2 Fuel Oil Acceptance Criteria; dated April 5, 2002

Completed PBF 3005; Blended #1 and #2 Fuel Oil Acceptance Criteria; dated December 28, 2004

Completed PBF 3005a; Quarterly Sampling of Emergency Fuel Oil Tanks – T-30 dated September 29, 2005

Completed PBF 3005a; Quarterly Sampling of Emergency Fuel Oil Tanks – T-32A; dated September 29, 2005

Completed PBF 3005a; Quarterly Sampling of Emergency Fuel Oil Tanks – T-32B; dated September 29, 2005

Completed RMP 9225-2; Defeating/Restoring Containment Personnel and Escape Hatch Door Interlocks; various from 2002 through 2005

Station Log; dated February 4, 2004

Station Log; dated February 6, 2004

TCN 2004-0339; Temporary Change - Isolation of the Containment Ventilation System Using the RMS High Alarm Automatic Trip Functions; dated May 21, 2004

Completed TS 10; Local Leak Test of Containment Airlock Bulkheads and Penetrations; dated March 27, 2005

Completed TS 10A; Containment Airlock Door Seal Testing Unit 2; dated March 31, 2005

Completed TS 80; Sampling of Emergency Fuel Oil Tanks (Quarterly); dated March 29, 2005

VPNPD 90-148; Supplement to 10 CFR 50.63, TAC. NOS. 68583 and 68587 Loss of All Alternating Current Power Point Beach Nuclear Plant, Unit 1 and 2; dated March 30, 1990

WEP-89-143; Letter from Westinghouse to Point Beach; Transmittal of Midloop Calculations; dated June 30, 1989

WO 9950688; P-38A AFP Mini Recirc Control; dated January 25, 2002

WO 9950689; P-38B AFP Mini Recirc Control; dated January 25, 2002

WO 9926779; Replace Equalizing Device in Accordance with MR 99-036*A; dated February 21, 2004

WO 9926780; Replace Equalizing Device in Accordance with MR 99-036*B; dated February 21, 2004

WO 0203762001; MOV Actuator Checkout; dated April 14, 2003

WO 0309001; Extend RH and SI Vent Lines per MR 02-011*B; dated October 7, 2005

WO 0403678; Inadvertent Letdown Isolation and Loss of Heaters (All Heaters Tripped Off) Control; June 17, 2004

LIST OF ACRONYMS USED

ADAMS	Agency-Wide Document Access and Management System
ASME	American Society of Mechanical Engineers
CCW	Component Cooling Water
CFR	Code of Federal Regulations
CVCS	Chemical and Volume Control System
DRP	Division of Reactor Projects
DRS	Division of Reactor Safety
EMA	Engineered Maintenance Action
IMC	Inspection Manual Chapter
IR	Inspection Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
PBNP	Point Beach Nuclear Plant
PRA	Probabilistic Risk Assessment
RCDT	Reactor Coolant Drain Tank
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
SBLC	Standby Liquid Control
SDP	Significance Determination Process
SPEED	Spare Parts Equivalency Evaluation Document
SSC	Structure, System, or Component
SW	Service Water
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item